ACCESSION #: 9908170018

NON-PUBLIC?: N

LICENSEE EVENT REPORT (LER)

FACILITY NAME: McGuire Nuclear Station, Unit 2 PAGE: 1 OF 8

DOCKET NUMBER: 05000370

TITLE: Reactor "Trip During SSPS Testing

EVENT DATE: 07-15-99 LER #: 99/04/0 REPORT DATE: 08/12/99

OTHER FACILITIES INVOLVED: Unit 2 DOCKET NO: 05000370

OPERATING MODE: 1 POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:

50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:

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COMPONENT FAILURE DESCRIPTION:

CAUSE: SYSTEM: COMPONENT: MANUFACTURER:

REPORTABLE NPRDS:

SUPPLEMENTAL REPORT EXPECTED:

ABSTRACT:

Unit Status: Unit 2 was in Mode 1 (Power Operation) at 100 percent power.

Event Description:

On July 15, 1999 at 14:44, an automatic turbine trip occurred on Unit 2 in response to an inadvertent reactor trip signal (P4). A turbine trip with the reactor power above 48 percent resulted in an automatic reactor trip. The P4 signal was generated during the replacement and testing of Reactor Trip Breaker A (RTA). SSPS Train A was in test alignment during the event; consequently and as designed Train A Auxiliary Feedwater

Pump did not automatically start and the RTA breaker did not automatically open. Operations personnel manually opened Train A reactor trip breaker and started the Train A Auxiliary Feedwater Pump.

Event Cause:

A 'switch lever arm' was misaligned during an inspection of a refurbished breaker. The breaker was subsequently located in the RTA cubicle: Inadequate procedural guidance during breaker receipt inspection is considered the primary cause for the misalignment of the lever arm. A contributing cause is inappropriate action. The misaligned lever arm resulted in an inadvertent P4 signal during subsequent installation testing.

Corrective Action:

Procedure enhancements will be made to clearly direct the actions to be taken when inspecting the reactor trip breakers. The installation testing procedure will be revised to improve the sequence of steps. Management reviewed expectations for procedure performance and work practices with appropriate maintenance crews emphasizing attention to detail.

NRC FORM 366 *_/NPRDS no longer exists, equipment failures will be reported through EPIX

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BACKGROUND:

The Solid State Protection System (SSPS) receives digital voltage inputs from the 7300 Process Control System, Nuclear Instrument System and other miscellaneous field inputs. The SSPS combines these signals in the required logic combination to generate actuation signals for automatic reactor trip and permissive logic for Engineered Safeguards actuation. In addition, the SSPS provides signals to control room annunciators [EIIS:ANN] and status lights [EIIS:IL] which indicate the condition of bistable input signals. Each bistable actuates at a given set point and initiates a signal through the Reactor Protection System (RPS) [EIIS:JC] or the

Engineered Safeguards System.

Reactor Trip Permissive (P4) is generated by an indication of the Reactor Trip Breaker Position. The signal is sent to the turbine trip controls to initiate a trip of the Main Turbine. The P4 signal is also used in the Steam Dump Control system, the protection logic for Feedwater isolation, and the reset circuitry for the Safety Injection signal. The Reactor Trip Breakers contain auxiliary contacts that mechanically close when the breaker opens. This closure completes the circuit for the P4 signal.

EVALUATION:

Description of Event

A refurbished breaker was received from Westinghouse on July 12, 1999 and was inspected per procedure SI/0/A/5100/002 (Westinghouse DS-416 Air Circuit Breaker Inspection And Maintenance). The breaker that had been located in the RTA cubicle was tested per procedure PT/0/A/4601/008 A (SSPS Train A Periodic Test With NC Pressure >1955 PSIG). The breaker from the RTA cubicle was scheduled to rotate to a Bypass Reactor Trip Breaker cubicle. The breaker from the bypass breaker location was next in line for refurbishing as part of a routine preventative maintenance rotation. On July 15, 1999, the refurbished breaker was installed in the RTA cubicle and tested per procedure PT/0/A/4601/008 A (SSPS Train A Periodic Test With NC System Pressure >1955 PSIG) to ensure proper operation. This procedure directed the technicians to measure resistance across auxiliary contacts to verify that the P4 Turbine Trip contact was open. On July 15, 1999 at

14:44, when the test instrument leads were placed on the contacts as directed by the procedure, the Main Turbine and the Reactor tripped. Primary and Secondary responses were as expected following the trip. Due to the testing configuration, the Train A Auxiliary Feedwater (EIIS:BA) Pump did not automatically start on receipt of a lolo Steam

Generator (EIIS:SG) level and the RTA breaker did not automatically open. Because Train A SSPS was in test, 'actuation logic' for A Train Engineered Safeguards Features and Reactor Protection System components which actuate on a P4 signal was blocked. The Train B Motor Driven and the Turbine Driven Auxiliary Feedwater Pumps automatically started as per design. Per existing procedural guidance Reactor Operators manually started the Train A Motor Driven pump and opened the Train A reactor trip breaker. Subsequent investigation indicates that the trip was primarily a result of action taken during the receipt of the refurbished breaker per SI/0/A/5100/002 (Westinghouse DS-416 Air Circuit Breaker Inspection And Maintenance). An auxiliary switch lever arm was found to be out of position. (This misalignment prevented the top auxiliary switch block from rotating, and thus changing state when the breaker was closed in the RTA cubicle.) As part of the inspection procedure, technicians were directed to measure the resistance between all auxiliary switch contacts. The procedure specified a maximum resistance limit of .5 ohms. Followup interviews indicate that the bottom switch had a measured resistance of 9 ohms. The appropriate procedure step directs the technicians to 'contact their work supervisor or appropriate engineer for further guidance, when exceeding the resistance acceptance criteria. The lead technician was qualified as a work supervisor and made the judgment that additional investigation was necessary to determine if the resistance reading was an accurate result.

In an effort to validate the ohmmeter reading, the technicians loosened the screws holding the auxiliary switches to the breaker and pulled back the switches to allow visual observation of the contacts. The ohmmeter was disconnected while the technicians were performing this visual inspection. The ohmmeter leads were reconnected at the original test points. A new measurement satisfied the acceptance criteria. The screws holding the switches were again torqued and the measured resistance value was recorded in the procedure.

The procedure was completed without re-verifying the correct alignment of the lever arm linkage. The breaker inspection procedure states that 'Inspection steps are NOT required to be performed in sequential order but are listed in a recommended sequence for ease of performance'. The technicians considered it acceptable to perform the troubleshooting and re-enter the procedure. During this visual inspection the lever arm linkage was still connected to the switches. It is postulated, that sometime during the visual inspection process the top auxiliary switch lever arm inadvertently rotated out of position and was reinstalled in this configuration. During testing following the reactor trip, it was determined that the auxiliary contact for the P4 signal for Turbine Trip

was closed when it should have been open. Checks earlier in the inspection procedure verify that the contacts were in the proper alignment on receipt. This limits the time frame in which the lever arm could have become misaligned to have most likely been during the visual inspection described above.

During SSPS testing, technicians were directed by procedure to measure resistance between points 4-2 and 4-3 (reference the drawing below) to verify an open circuit across auxiliary contact 52b BYA. Post trip testing confirmed that the ohmmeter could provide the required current path to the turbine trip circuitry with the 52b RTA contact closed. The same ohmmeter used at the time of the trip was connected to terminals 42 and 4-3. The ohmmeter was found to function properly and was found to be capable of passing adequate current to close the remaining contacts for the P4 turbine trip circuitry. Refurbished RTA was then removed from the RTA cubicle and inspection revealed that the top auxiliary switch lever arm was out of position. This prevented the top auxiliary switch block from rotating properly when Breaker RTA was closed.

Figure : "125 VDC Turbine Trip Circuitry, 125 VDC Turbine Trip Circuitry" omitted.

NOTE: 52b contacts close when the breaker opens and open when the breaker closes.

33b contacts close when the breaker is racked out and open when

the breaker is racked to connect.

The breaker that had been in the RTA cubicle was reinstalled in the RTA cubicle and the Train A SSPS testing was completed to restore SSPS operability. The refurbished breaker was removed and the top auxiliary switch was found out of position as described above.

Conclusion

o This event did not result in any uncontrolled releases of radioactive material, personnel injuries, or radiation overexposures. This event would not have been Nuclear Plant Reliability Data System (NPRDS) reportable. NPRDS has been replaced by the Equipment Performance Information Exchange (EPIX).

Causal Evaluation

Inadequate procedural guidance during breaker receipt inspection is considered the primary cause for the misalignment of the lever arm. In particular, the number and complexity of steps that can be performed out of sequence is not appropriate. This is also considered the primary cause of the subsequent reactor trip on turbine trip. Inappropriate action associated with the visual inspection process is considered a contributing factor to the lever arm misalignment. In particular, there was not adequate attention to detail in reviewing the condition of the breaker following the visual inspection. The misalignment of the lever arm is a visible condition. As an additional contributing factor, the SSPS testing procedure was found to be inadequate. In particular, the procedure did not require a preliminary voltage check prior to the resistance reading. The voltage check would have indicated that the P-4 contact was closed when it should have been open. In addition, there were no steps requiring a continuity check across the 4-1 and 4-4 contacts when the main (RTA) breaker was racked in and closed on the bus.

Operating Experience

A review of the Operating Experience Program (OEP) and Problem Investigation Process (PIP) databases for the past 24 months revealed one item associated with the P-4 contacts on the Reactor Trip Breakers. This item was associated with the type of lubricant used on the contacts and was not related to problems identified during this event. The review revealed one event of inadequate SSPS testing caused by inadequate test design supplied by the vendor. Inadequate development and review of a procedure change caused one event of an ESF actuation associated with the Diesel Generator Load Sequencer. This event is considered to be recurring due to the procedural problem associated with an ESF actuation. However, the procedures involved different ESF features, procedure writers, and work crews. Given the large number of

ESF procedures and work activities, this recurrence does not warrant broader corrective actions.

CORRECTIVE ACTION:

Immediate

Control room personnel entered procedure EP/2/A/5000/E-0 (Reactor Trip Or Safety Injection) and EP/2/A 5000/ES-0.1 (Reactor Trip Response).

Subsequent

1. Procedure PT/0/A/4700/45 (Reactor Trip Investigation) was entered to address and resolve issues following the reactor trip.

2. Testing was performed on the refurbished Breaker in the RTA cubicle to determine the cause of the trip.

3. The breaker that had been in the RTA cubicle was reinstalled in the RTA cubicle and SSPS testing was completed to restore SSPS operability.

4. Maintenance management has communicated the details of this event to appropriate maintenance crews. This communication focused on the importance of attention to detail.

5. Procedure PT/0/A/4601/008 A,B (SSPS Train A,B Periodic Test With NC System Pressure >1955 PSIG) was revised to require voltage checks in place of resistance checks where the ohmmeter could complete a current path. These voltage checks will detect a contact that is not in its expected condition.

Planned

1. Procedure SI/0/A/5100/002 (Westinghouse DS-416 Air Circuit Breaker

Inspection And Maintenance) will be revised to give specific guidance for the inspection activities including the sequence of the steps.

SAFETY ANALYSIS:

This event is not considered to be significant. At no time were the safety or health of the public or plant personnel affected as a result of the event.

A transient caused by a turbine trip is analyzed in chapter 15 of the McGuire Nuclear Station Final Safety Analysis Report. The accident analysis demonstrates that the plant design is such that a turbine trip presents no hazard to the integrity of the reactor coolant system or the main steam system. Pressure relieving devices incorporated in the two systems are adequate to limit the maximum pressures within design limits.

The unit responded to the Turbine Trip and Reactor Trip as designed for the existing conditions. The Train A Solid State Protection System was in the test alignment. Protection signals, including the automatic start of the Train A Motor Driven Auxiliary Feedwater Pump and the trip signal to the RTA breaker, generated by the Train A SSPS are blocked while in the test alignment. The Train B SSPS generated a Reactor Trip due to a Turbine Trip with the Unit above 48 percent power. The Train B Motor Driven Auxiliary Feedwater Pump and the Turbine Driven Auxiliary Feedwater Pump automatically started as required. The Train A Auxiliary Feedwater Pump was started by the Reactor Operators. Steam Generator Levels were at their approximate no-load values 30 minutes after the trip. Main Reactor Trip Breakers 2RTB, and bypass breakers 2BYB and 2BYA opened automatically upon receipt of a Reactor Trip signal. The Reactor Operator manually opened breaker 2RTA as directed by procedure. Adequate core cooling was maintained throughout the transient and the Reactor Coolant System pressure boundary was not challenged. The actual event is bounded by the results of the existing accident analysis.

The misaligned linkage in the reactor trip breaker would not have adversely affected the safety function of the breaker. The reactor trip breaker would have actuated to trip the reactor on receipt of a reactor trip signal.

The core damage probability significance of this event has been evaluated by considering the following.

o A turbine trip initiating event.

o Potential for operator action to control the reactor trip breaker and CA pump that were affected by the testing.

o Actual plant configuration at the time of the trip.

This event was far less severe than a design basis turbine trip event. The inability of one reactor trip breaker and one motor-driven CA pump to function automatically has little impact on the overall sequence quantification. The conditional core damage probability for the event evaluated is low, < 4E-07. This event does not represent an accident sequence precursor.

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